NEUTRONS

Production

The most prolific source of neutrons is the nuclear reactor. The splitting of a uranium or a plutonium nucleus in a nuclear reactor is accompanied by the emission of several neutrons. These fission neutrons have a wide range of energies, as shown in Fig(4-1). The distribution peaks at 0.7 MeV and has a mean value of 2 MeV .Except for several fission fragments of very short half-life, there are no radionuclides that decay by emitting neutrons.

Californium-252, however, an alpha emitter, undergoes spontaneous nuclear fission at an average rate of 10 fissions for every 313 alpha transformations. Since the half-life of ^{252}Cf due to alpha emission is 2.73 years, its effective half-life, including spontaneous nuclear fission, is 2.65 years. Californium-252 thus simulates a neutron-emitting radionuclide. The neutron emission rate has been found to be 2.31×10^6 neutrons per second per μg ^{252}Cf.

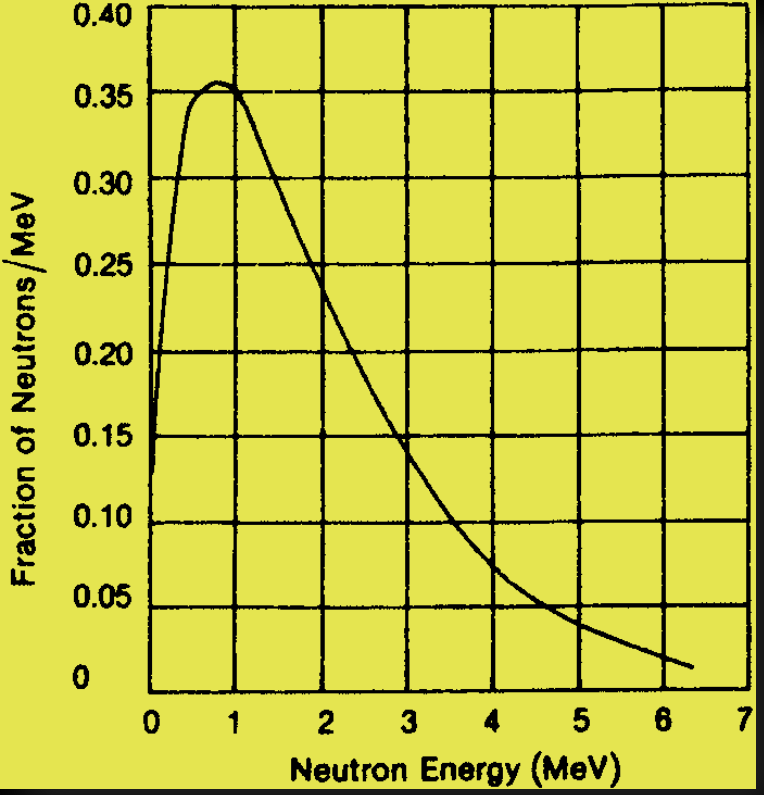

Figure 4-1. Energy distribution of fission ns. The most probable energy is 0.7 MeV and the average energy is 2 MeV.

The n emission rate has been found to be 2.31×10^{6} ns $S^{-1}\mu g^{-1}$ ^{252}Cf. The emitted ns span a wide range of energies.The most probable energy is about 1 MeV, while the average value of the energy distribution is about 2.3 MeV.

All other neutron sources depend on nuclear reactions for the emission of neutrons. Copious neutron beams may be produced in accelerators by many different reactions. For example, bombardment of beryllium by high-energy deuterons in a cyclotron produces neutrons according to the reaction

$$ {}^{9}_{4}\mathrm{Be} + {}^{2}_{1}\mathrm{D} \rightarrow \left({}^{11}_{5}\mathrm{B}\right)^{*} \rightarrow {}^{10}_{5}\mathrm{B} + {}^{1}_{0}\mathrm{n}. \qquad (4\text{-}1) $$

The term in the parentheses is called a compound nucleus, and the asterisk shows that it is in an excited state.

The compound nucleus rids itself of its excitation energy instantaneously (<10−8 s) by proceeding to the next step in the reaction. For small laboratory sources of neutrons, the photodisintegration of beryllium may be used. Another commonly used neutron source depends on the bombardment of beryllium with alpha particles. The reaction, in this case, is

$$^{9}_{4}\mathrm{Be} + {}^{4}_{2}\mathrm{He} \rightarrow \left({}^{13}_{6}\mathrm{C}\right)^{*} \rightarrow {}^{12}_{6}\mathrm{C} + {}^{1}_{0}\mathrm{n}. \qquad (4\text{-}2)$$

For the source of the alpha particles, radium, polonium, or plutonium is used. The α emitter, as a powder, is thoroughly mixed with finely powdered beryllium, and the mixture is sealed in a capsule, as shown in Fig 4-2. The ns that are produced are all high-energy ns. In all cases of ns based on this reaction, the n energy is spread over a broad spectrum (Fig 4-3). This spread of energies from a $^{9}Be(\alpha, n)^{12}C$ source is in sharp contrast to the monoenergetic neutrons from a photodisintegration source using monoenergetic photons.

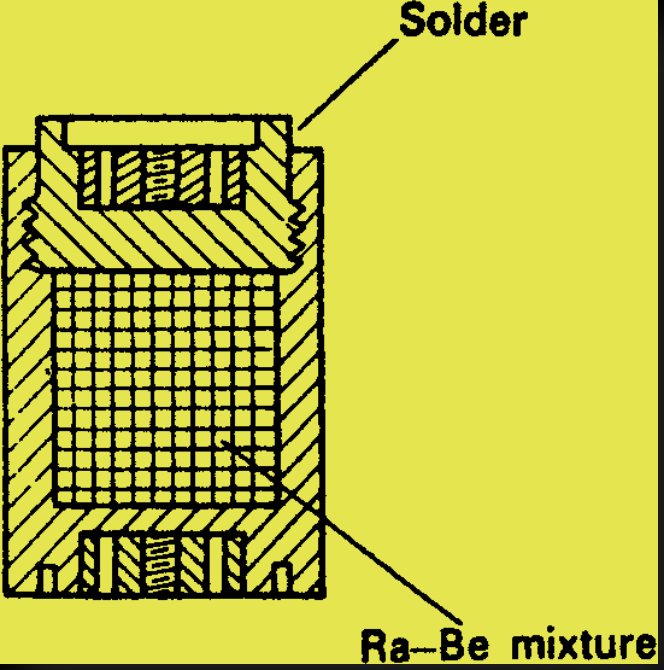

Figure 4-2. Typical construction of an α, n neutron source in a sealed container. Here Ra serves as the α source.

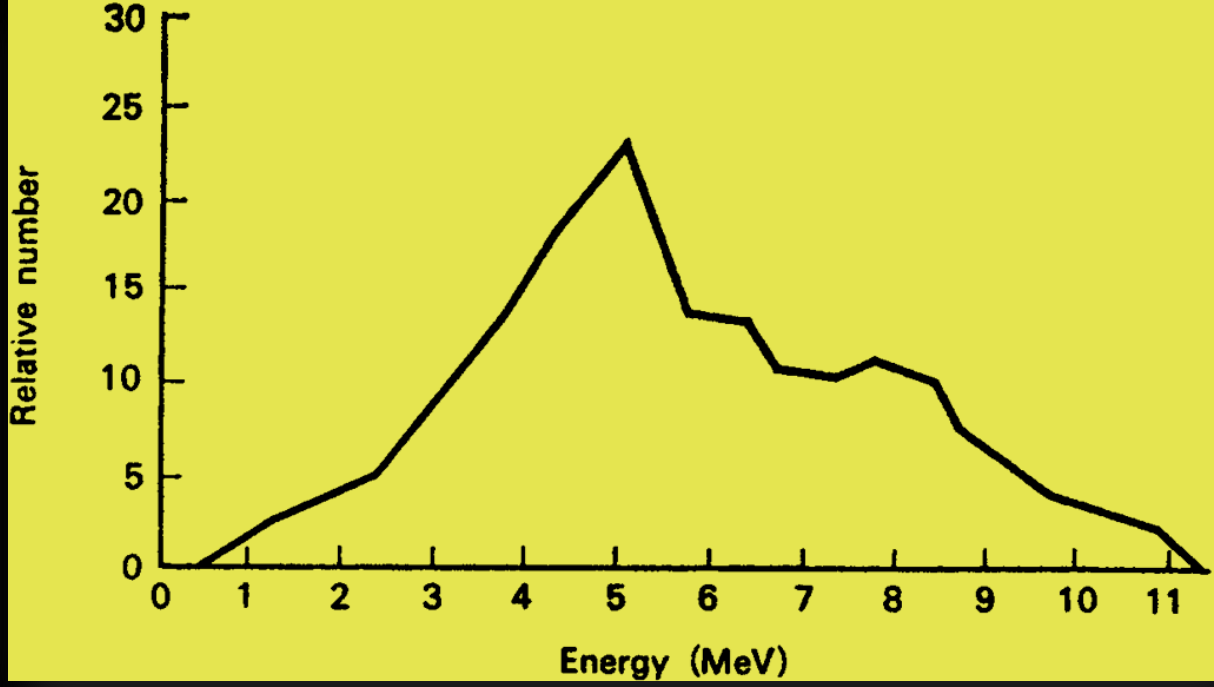

Figure 4-3. Energy distribution of Po–Be neutrons.

In the α, n reaction, the energy equivalent of the difference in mass between the reactants and the products plus the K.E.bombarding particle is divided between the n and the recoil nucleus. In practical α, n sources, some of the α- energy is dissipated by self-absorption within the source .So the αs that initiate the reaction have a wide range of energies, illustrating the spectral spread of the ns. The n yield for an α, n source increases with increasing α energy because of the greater ease with which higher-energy αs penetrate the coulomb barrier at the nucleus.

Tables 4-1 and 4-2 list some γ , n and α, n neutron sources, respectively.

γ , n Photoneutron Sources

SOURCE	HALF-LIFE	AVERAGE NEUTRON ENERGY (MeV)	YIELD $\frac{n}{s}$/Ci	YIELD $\frac{n}{s}$/MBq
$^{24}Na + Be$	15 h	0.83	1.3×10^5	3.5
$^{24}Na + D_2O$	15 h	0.22	2.7×10^5	7.3
$^{56}Mn + Be$	2.58 h	0.1(90%), 0.3(10%)	2.9×10^4	0.8
$^{56}Mn + D_2O$	2.58 h	0.22	3.1×10^3	0.08
$^{72}Ga + Be$	14.2 h	0.78	5×10^4	1.4
$^{72}Ga + D_2O$	14.2 h	0.13	6×10^4	1.6
$^{88}Y + Be$	88 d	0.16	1×10^5	2.7
$^{88}Y + D$	88 d	0.31	3×10^3	0.08
$^{116}In + Be$	54 min	0.30	8.2×10^3	0.2
$^{124}Sb + Be$	60 d	0.024	1.9×10^5	5.1
$^{140}La + Be$	40 h	0.62	3×10^3	0.08
$^{140}La + D_2O$	40 h	0.15	8×10^3	0.2
$Ra + D_2O$	1600 yrs	0.12	1×10^3	0.03

α, n Neutron Sources

SOURCE	HALF-LIFE	AVERAGE NEUTRON ENERGY (MeV)	YIELD	
			$\frac{n}{s}$/Ci	$\frac{n}{s}$/MBq
Ra + Be	1600 yrs	5	1.7×10^7	459
Ra + B	1600 yrs	3	6.8×10^6	184
^{222}Rn + Be	3.8 d	5	1.5×10^7	405
^{210}Po + Be	138 d	4	3×10^6	81.1
^{210}Po + B	138 d	2.5	9×10^5	24.3
^{210}Po + F	138 d	1.4	4×10^5	10.8
^{210}Po + Li	138 d	0.42	9×10^4	2.4
^{239}Pu + Be	24,000 yrs	4	10^6	27

Classification

Neutrons are classified according to their energy because the type of reaction that a neutron undergoes depends very strongly on its energy. High-energy neutrons, those whose energies exceed about 0.1 MeV, are called fast neutrons. Thermal neutrons, on the other hand, have the same kinetic energy distribution as gas molecules in their environment. In this respect, thermal neutrons are indistinguishable from gas molecules at the same temperature.

The kinetic energies of gas molecules are related to temperature by the Maxwell–Boltzmann distribution (Fig. 4-4) :

$$f(E) = \frac{2\pi}{(\pi kT)^{3/2}} e^{-E/kT} E^{1/2} \quad (4\text{-}3)$$

where f (E) is the fraction of the gas molecules (or neutrons) of energy E per unit energy interval; k is the Boltzmann constant, 1.38×10^{-23} J/K or 8.6×10^{-5} eV/K; and T is the absolute temperature of the gas, K (Kelvin).

The most probable energy, represented by the peak of the curve in Figure 4-4, is given by

$$E_{\mathrm{mp}} = kT, \qquad (4\text{-}3)$$

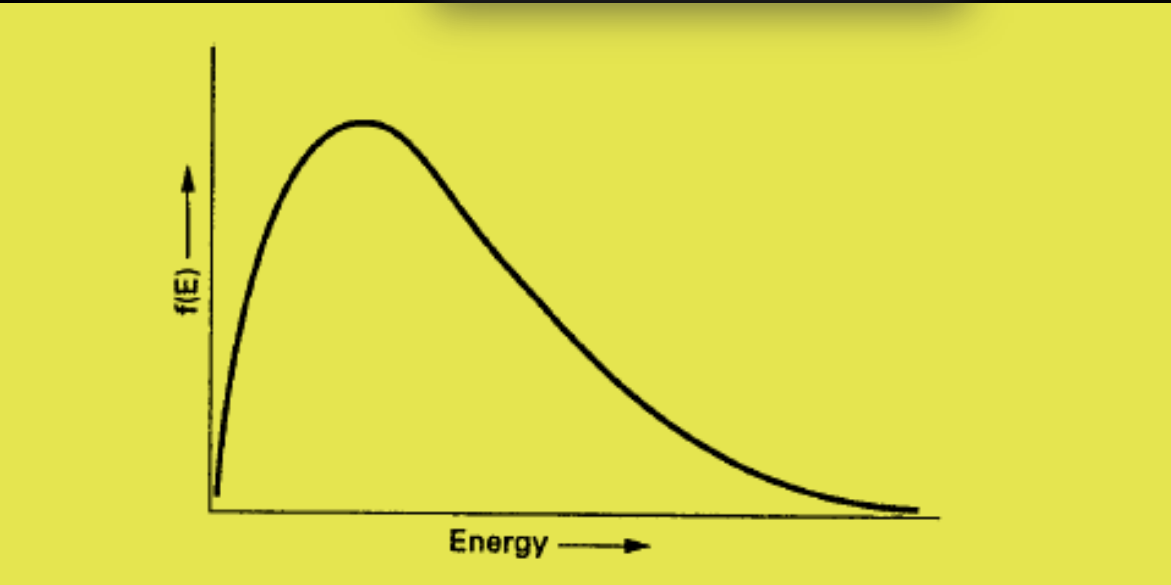

Figure 4-4.Maxwell –Boltzmann distribution of energy among gas molecules.

the average energy of gas molecules at any given temperature is

$$\bar{E} = \frac{3}{2}kT. \qquad (4\text{-}4)$$

For neutrons at a temperature of 293 K, the most probable energy is 0.025 eV. This is the energy often implied by the term "thermal" neutrons. The velocity corresponding to this energy, is calculated as

$$\frac{1}{2}mv^2 = kT = 0.025\,\text{eV} \times 1.6 \times 10^{-19}\ \frac{\text{J}}{\text{eV}}$$

$$v = \left(\frac{2 \times 0.025\,\text{eV} \times 1.6 \times 10^{-19}\,\text{J/eV}}{1.67 \times 10^{-27}\text{kg}}\right)^{1/2} = 2.2 \times 10^3\,\frac{\text{m}}{\text{s}}.$$

The average velocity of a neutron in a thermal n beam, if the most probable velocity is v_0, is given by

$$\overline{v} = \frac{2}{\sqrt{\pi}} v_0 = 1.13\ v_0. \qquad (4\text{-}5)$$

In the region of energy between thermal and fast, neutrons are called by various names including intermediate neutrons, resonance neutrons, and slow neutrons. All these descriptive adjectives are used loosely, and their exact meaning must be inferred from the context in which they are used.

Interaction

All neutrons at the time of their birth are fast. Generally, fast ns lose energy by colliding elastically with atoms in their environment, and then, generally after being slowed down to thermal or near thermal energies, they are captured by nuclei of the absorbing material. Although a number of possible neutron reaction types exist, the chief reactions for the health physicist are elastic scattering and capture followed by the emission of a photon or another particle from the absorber nucleus.

When absorbers are placed in a collimated beam of ns and the transmitted neutron intensity is measured, as was done for γ rays in Fig 3-1, it is found that ns too are removed exponentially from the beam. Instead of using linear or mass absorption coeffs to describe the ability of a given absorber material to remove ns from the beam, it is customary to designate only the microscopic cross section σ for the absorbing material. The product σ N, where N is the number of absorber atoms per cm^3, is the macroscopic cross section Σ .

The removal of neutrons from the beam is thus given by

$$I = I_0 e^{-\sigma N t}. \qquad (4\text{-}6)$$

Neutron cross sections are strongly energy dependent. If removal of a n from the beam may be effected by more than one mechanism, the total cross section is the sum of the cross sections for the various possible reactions.

EXAMPLE 4-1

In an experiment designed to measure the total cross section of lead for 10-MeV ns, it was found that a 1cm-thick lead absorber attenuated the neutron flux to 84.5% of its initial value. The atomic weight of lead is 207.21 and its specific gravity is 11.3. Calculate the total cross section from these data.

Solution

The atomic density of lead is

$$\rho_{\text{atomic}} = \frac{6.02 \times 10^{23}\ \text{atoms/mol}}{207.21\ \frac{\text{g}}{\text{mol}}} \times 11.3\ \frac{\text{g}}{\text{cm}^3} = 3.3 \times 10^{22}\ \frac{\text{atoms}}{\text{cm}^3}$$

$$\frac{I}{I_0} = e^{-\sigma N t}$$

$$0.845 = e^{-\sigma \times 3.3 \times 10^{22} \times 1}$$

$$\ln \frac{1}{0.845} = 3.3 \times 10^{22} \sigma$$

$$\sigma = \frac{0.168}{3.3 \times 10^{22}\ \frac{\text{atoms}}{\text{cm}^3} \times 1\ \text{cm}} = 5.1 \times 10^{-24} \frac{\text{cm}^2}{\text{atom}}.$$

$\sigma = 5.1$ b and the macroscopic cross section is

$$\sum = \sigma N = 5.1 \times 10^{-24}\ \text{cm}^2/\text{atom} \times 3.3 \times 10^{22}\ \text{atoms/cm}^3 = 0.168\ \text{cm}^{-1}.$$

Scattering

Neutrons may collide with nuclei and undergo either inelastic or elastic scattering. In the former case, some of the kinetic energy that is transferred to the target nucleus excites the nucleus and the excitation energy is emitted as a gamma-ray (photon).This interaction is best described by the compound nucleus model in which the neutron is captured and then reemitted by that target nucleus together with the gamma ray (photon).

This is a threshold phenomenon; the neutron energy threshold varies from infinity for hydrogen (inelastic scattering cannot occur) to about 6 MeV for oxygen to less than 1 MeV for uranium. Generally, the cross section for inelastic scattering is small (on the order of 1 b or less) for low-energy fast neutrons, but it increases with increasing energy and approaches a value corresponding to the geometric cross section of the target nucleus .

Elastic scattering is the most likely interaction between fast neutrons and low atomic-numbered absorbers. This interaction is a "billiard-ball" type collision, in which kinetic energy and momentum are conserved. By applying these conservation laws, it can be shown that the energy E of the scattered neutron after a head-on collision is

$$E = E_0 \left\{ \frac{M - m}{M + m} \right\}^2 \qquad (4\text{-}7)$$

where

E_0 = energy of the incident neutron,
m = mass of the incident neutron, and
M = mass of the scattering nucleus.

The energy transferred to the target nucleus is $E_0 - E$. From Eq. (4-7), one can get

$$E_0 - E = E_0 \left[1 - \left(\frac{M - m}{M + m} \right)^2 \right]. \qquad (4\text{-}8)$$

According to Eqs. (4-7) and (4-8), it is possible, in a head-on collision with a hydrogen nucleus, for a neutron to transfer all its energy to the hydrogen nucleus. With heavier nuclei, all the kinetic energy of the neutron cannot be transferred in a single collision.

In the case of oxygen, for ex, Eq. (4-8) shows that the maximum fraction, $(E_0 - E) = E_0$, of the neutron's kinetic energy that can be transferred during a single collision is only 22.2%. This shows that nuclei with small mass numbers are more effective, on a (per collision) basis, than nuclei with high As for slowing down neutrons. Eqs (4-7) and (4-8) are valid only for head-on collisions. Most collisions are not head-on, and the energy transferred to the target nuclei are consequently less than the maximum given by the two equations above.

In the course of the successive collisions suffered by a fast neutron as it passes through a slowing-down medium, the average decrease, per collision, in the logarithm of the n energy (which is called the average logarithmic energy decrement) remains constant. It is independent of the neutron energy and is a function only of the mass of scattering nuclei. The average logarithmic energy decrement is defined as

$$\xi = \overline{\Delta \ln E} = \overline{\ln E_0 - \ln E} = \overline{\ln \frac{E_0}{E}} = \overline{-\ln \frac{E}{E_0}} \qquad \textit{(4-9)}$$

and can be shown to be given by

$$\xi = 1 + \frac{\alpha \ln \alpha}{1 - \alpha} \qquad \textit{(4-10)}$$

where $\alpha = [(M - m)/(M + m)]^2$ (4-11)

$$\xi = \frac{\sum_{i=1}^{n} \sigma_{si} N_i \xi_i}{\sum_{i=1}^{n} \sigma_{si} N_i} \qquad (4\text{-}12)$$

since

$$\overline{\ln \frac{E}{E_0}} = -\xi,$$

$$\frac{E}{E_0} = e^{-\xi},$$

and the median fraction of the incident neutron's energy that is transferred to the target nucleus during a collision is

$$f = 1 - \frac{E}{E_0} = 1 - e^{-\xi}. \qquad (4\text{-}13)$$

Thus, for hydrogen ($\xi = 1$), the median energy transfer during a collision with a fast n is 63% of the n's K.E. In the case of carbon, $\xi = 0.159$, and an average of only 14.7% of the n's K.E.is transferred to the struck nucleus during an elastic collision. The struck nucleus, as a result of the K.E.imparted to it by the n, becomes an ionizing particle and dissipates itsK.E.in the medium by excitation and ionization.

The distance traveled by a fast neutron between its introduction into a slowing down medium and its thermalization depends on the number of collisions made by the neutron and the distance between collisions. Although the actual path of the neutron is tortuous because of deflections due to collisions, the average straight line distance covered by the neutron can be determined; it is called the fast-diffusion length, or the slowing-down length

(The square of the fast-diffusion length is called the Fermi age of the neutron.) The distance traveled by the thermalized neutron until it is absorbed, is measured by the thermal diffusion length. The thermal diffusion length is defined as the thickness of a slowing-down medium that attenuates a beam of thermal ns by a factor of e . Thus, attenuation of a beam of thermal ns by a substance of thickness t cm whose thermal diffusion length is L cm is given by

$$n = n_0 e^{-t/L}. \qquad (4\text{-}14)$$

(The terms fast diffusion length and thermal diffusion length are applicable only to materials in which the absorption cross section is very small. When this condition is not met, as in the case of boron or cadmium, the attenuation of a beam of thermal ns is given by Eq.4-6). Although fast and thermal diffusion lengths may be calculated. Calculations make it preferable to use measured values for these parameters. Values for fast and thermal diffusion lengths for fission neutrons in certain slowing-down media are given in Table 4-3.

TABLE 4-3. Fast and Thermal Diffusion Lengths of Selected Materials

SUBSTANCE	FAST DIFFUSION LENGTH (cm)	THERMAL DIFFUSION LENGTH (cm)	DIFFUSION COEFFICIENT (cm)
H_2O	5.75	2.88	0.16
D_2O	11	171	0.87
Be	9.9	24	0.50
C (graphite)	17.3	50	0.84

If a point source of S thermal ns per sec in a spherically shaped nonmultiplying medium (a medium which contains no fissile material) of radius R, thermal diffusion length L, and diffusion coeff D, the flux of ns escaping from the surface is

$$\phi = \frac{S}{4\pi RD} \times e^{-R/L}. \qquad (4\text{-}15)$$

EXAMPLE (4-2)

A Pu–Be neutron source that emits 10^6 n/s is in the center of a spherical water shield whose diameter is 50 cm. How many thermal neutrons are escaping /cm^2 /sec from the surface of the shield?

Since the radius of the water shield is much greater than the fast diffusion length given in Table 4-3, we may assume (for the purpose of this calculation) that essentially all the fast ns are thermalized and that the thermal ns are diffusing outward from the center. Substituting the appropriate numbers into Eq. (4-15), one get

$$\phi = \frac{10^6 \text{ neutrons/s}}{4\pi \times 25 \text{ cm} \times 0.16 \text{ cm}} e^{-25 \text{ cm}/2.88 \text{ cm}} = 3.4 \frac{\text{neutrons/s}}{\text{cm}^2}.$$

Absorption

The discussion above shows that fast ns are rapidly degraded in energy by elastic collisions if they interact with low Z substances. As ns reach thermal or near thermal energies, their likelihood of capture by an absorber nucleus increases. σ of many nuclei, as the n energy becomes very small, has been found to be inversely proportional to the square root of its K.E.and thus to vary inversely with its velocity:

$$\sigma \propto \frac{1}{\sqrt{E}} \propto \frac{1}{v}. \qquad (4\text{-}16)$$

Eq (4-17) is called the one-over-v law for slow neutron absorption. For ^{10}B, this relationship is valid for the span of energies from 0.02 to 1000 eV, as shown in fig 4-5. Thermal neutron cross sections are usually given for neutrons whose most probable energy is 0.025 eV. If the cross section at energy E_0 is σ_0, then the cross section for any other energy within the range of validity of the 1/v law is given by

$$\frac{\sigma}{\sigma_0} = \frac{v_0}{v} = \sqrt{\frac{E_0}{E}}. \qquad (4\text{-}18a)$$

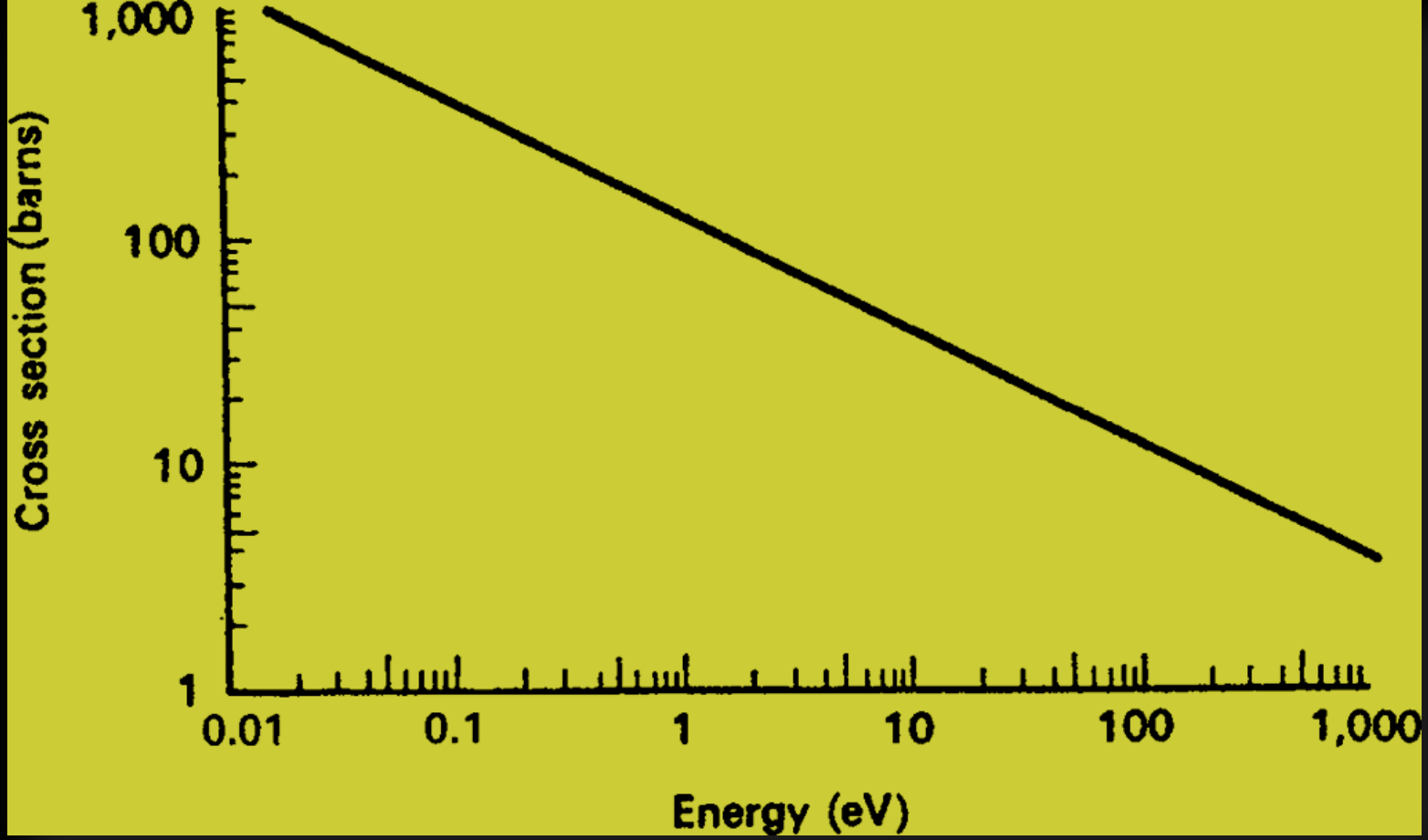

Figure 4-5. Neutron absorption cross sections for boron, showing the validity of the 1/v law for neutrons from 0.02 to 1000 eV in energy. The equation for the curve is $\sigma = 116\ (eV)^{1/2}$.

Since neutron energy varies directly with the temperature, Eq. (4-18a) can be written as

$$\frac{\sigma}{\sigma_0} = \frac{v_0}{v} = \sqrt{\frac{T_0}{T}}, \qquad \textit{(4-18b)}$$

where $T_0 = 293$ K.

EXAMPLE 4-3

The cross section of boron for the $^{10}B(n, \alpha)^{7}Li$ reaction is 753 b for 0.025 eV neutrons. What is the boron cross section for 50-eV neutrons?

Solution

Substituting into Eq. (4-18a) gives

$$\sigma\ (50\ \text{eV}) = 753\ \text{b}\sqrt{\frac{0.025\ \text{eV}}{50\ \text{eV}}} = 16.8\ \text{b}.$$

Some capture reactions of practical importance in health physics include the following:

$^{1}\text{H}(n, \gamma)^{2}\text{H}$	$\sigma = 0.33\,\text{b}$
$^{14}\text{N}(n, p)^{14}\text{C}$	$\sigma = 1.70\,\text{b}$
$^{10}\text{B}(n, \alpha)^{7}\text{Li}$	$\sigma = 4.01 \times 10^{3}\,\text{b}$
$^{113}\text{Cd}(n, \gamma)^{114}\text{Cd}$	$\sigma = 2.1 \times 10^{4}\,\text{b}$

(4-19…22)

Eqs (4-19) and (4-22) are important in n dosimetry, since H and N are major constituents of tissue. Eq. (4-21) is important in the design of instruments for measuring ns as well as neutron shielding, while the last equation is important mainly in shielding. It should be noted that the n reactions with H and with Cd result in the emission of high-energy γ rays, while the capture of a thermal neutron by ^{10}B releases a low-energy (0.48 MeV) γ ray in 93% of the reactions. When a thermal neutron is captured by ^{14}N, a 0.6-MeV proton is emitted.

Neutron Activation

Neutron activation is the production of a radionuclide by absorption of a neutron, such as the n, p reaction of Eq. (4-20). In that instance, ^{14}C is produced. Activation by neutrons is important to the health physicist for several reasons. First, it means that any substance that was irradiated by neutrons may be radioactive; a radiation hazard may therefore persist after the irradiation by neutrons is terminated. Secondly, it provides a convenient tool for measuring neutron flux.

This is done simply by irradiating a known quantity of the material to be activated, measuring the induced activity, and then, with a knowledge of the activation cross section, computing the neutron flux. In case of a criticality accident (an accidental attainment of an uncontrolled chain reaction), the measurement of induced radioactivity due to neutron irradiation permits calculation of the neutron dose. The same principle is applied by the chemist in neutron-activation analysis.

This method, which for many elements is more sensitive than other physical or chemical procedures, involves irradiation of the unknown sample in a neutron field of known intensity and radio spectrometric examination of the induced radiation to identify the isotope, which, in turn, helps to identify the unknown isotope from which it came and the amount of the unknown in the sample.

If a radionuclide is being made by neutron irradiation and is decaying at the same time, the net number of radioactive atoms present in the sample at any time is the difference between the rate of production and the rate of decay. This may be expressed mathematically by the following activity–balance equation:

net rate of increase of radioactive atoms

= production rate − decay rate,

that is,

$$\frac{\mathrm{d}N}{\mathrm{d}t} = \phi\sigma n - \lambda N, \qquad (4\text{-}23)$$

where

ϕ = flux, neutrons/cm^2/s,
σ = activation cross section, cm^2,
λ = transformation constant of the induced activity,
N = number of radioactive atoms, and
n = number of target atoms (assumed to remain constant during the irradiation).

$$\lambda N = \phi \sigma n \left(1 - e^{-\lambda t}\right). \qquad (4\text{-}24)$$

In Eq. (4-24), φσn is sometimes called the saturation activity. For an infinitely long irradiation time, it represents the maximum obtainable activity with any given neutron flux.

EXAMPLE 4-4

A sample containing an unknown quantity of Cr is irradiated for 1 week in a thermal n flux of 10^{11} n/cm^2/s. The resulting ^{51}Cr γ rays give a counting rate of 600 counts/min in a scintillation counter whose overall efficiency is 10%. How many grams of Cr were there in the sample?

Solution

The reaction in this case is

$$^{50}\mathrm{Cr} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{51}\mathrm{Cr} + \gamma.$$

The thermal neutron activation cross section for ^{50}Cr is 13.5 b, and 50Cr forms 4.31% by number of the naturally occurring Cr atoms. Cr-51 decays by orbital electron capture with a half-life of 27.8 days and emits a 0.323-MeV γ ray in 9.8% of the decays. The atomic weight of Cr is 52.01.The activity is given by λN in Eq.4-24. This equation may therefore be solved for n, the number of target atoms. Substituting the numerical values into Eq. 4-24

$$10\frac{\text{counts}}{\text{s}} \times 10\frac{\text{decays}}{\text{count}} = 10^{11}\frac{\text{neutrons}}{\text{cm}^2 \cdot \text{s}} \times 1.35$$

$$\times\ 10^{-23}\frac{\text{cm}^2}{\text{atom}} \times 0.098 \times n \text{ atoms } {}^{50}\text{Cr} \times \left(1 - e^{-\frac{0.693}{27.8\text{ d}} \times 7\text{ d}}\right)$$

$$n = 4.7 \times 10^{15} \text{ atoms } {}^{50}\text{Cr}.$$

Since the abundance of ^{50}Cr atoms is 4.31%, the total number of Cr atoms in the sample is

$$\text{No. Cr atoms} = \frac{n}{0.0431} = \frac{4.7 \times 10^{15}}{0.0431} = 1.1 \times 10^{17}\text{Cr atoms}$$

Since there are 52.01 g Cr/mol, the weight of chromium in the sample is

$$\frac{1.1 \times 10^{17} \text{ atoms}}{6.02 \times 10^{23} \text{ atoms/mol}} \times 52.01\ \frac{\text{g}}{\text{mol}} = 9.5 \times 10^{-6}\text{ g}.$$

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